APPENDIX C

DESCRIPTION OF SRP HIGH-LEVEL WASTE

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Chemical separations of irradiated fuel and targets at SRP result in product streams and acidic waste streams that contain almost all of the fission products, and small amounts of unrecovered uranium and transuranics. This acidic waste stream is made alkaline (pH 10 to 13) by addition of sodium hydroxide and transferred to large (about 4,900 m^3) underground storage tanks with multiple barriers of carbon steel and reinforced concrete.

In the waste storage tanks, components insoluble in the highly alkaline solution precipitate and settle to form a layer of flocculent sludge on the tank bottom. Most of the radioactive elements, including strontium and the actinides, are contained in the sludge; only the cesium remains predominantly in the supernatant liquid. Settled sludge volume is from 4 to 7% of the waste received, but 70 to 90% of this volume is interstitial liquid with a composition similar to the supernate.

After one to two years' storage, radioactivity of short-lived fission products has largely decayed, and the diminished thermal agitation permits most of the suspended sludge and associated radioactive components to settle out. Then the supernatant liquid, containing most of the soluble, nonradioactive salts and the radioactive cesium, is decanted off to other waste tanks and processed through evaporators to remove most of the water.

The partially dewatered waste concentrate from the evaporators is discharged to waste tanks while hot. On cooling, part of the dissolved salt mixture (chiefly sodium nitrate, nitrite, carbonate, sulfate, and hydroxide) crystallizes out of solution and deposits in the tank as damp salt cake. The remaining supernatant liquid is recycled back to the evaporators for removal of more water and additional crystallization of salt cake.

About 110,000 m^3 (28 million gallons) of high-level waste are presently stored at SRP. The actual volumes at any time in the future will be a function of the waste generation from plant operations, DWPF startup, and the operations to concentrate the waste.

The sludge (containing most of the strontium-90 and the actinides) will be the initial feed to the DWPF. High-activity components from the supernate (primarily cesium-137 and small amounts of strontium and the actinides) will be concentrated in another facility for mixing with the sludge feed to the DWPF or recovered for beneficial use. Continued development of supernate processing technology is expected to reduce significantly the cost and complexity of the supernate decontamination and disposal process described in the DWPF EIS.[1]

The sludge characteristics will determine the composition and properties of the waste form. The composition of the existing sludge varies considerably from tank to tank and, to a lesser extent, within each tank. Principal elements of the sludge measured from samples taken from several tanks are listed in Table C-1. The effects of waste composition on glass product performance have been studied with simulated waste glass,[2] and acceptable performance has been obtained with compositional variations more extreme than expected in practice.

Heat Generation

SRP waste storage tanks now contain about 2.6 MW of heat-generating fission products. The major contributors are:

Isotope	Megawatts	Half-Life, yr
Cs-137	0.63	30.0
Sr-90	0.77	28.0
Ce-144	0.63	0.78
Misc.	0.53	-

Without replenishment from fresh waste, heat generation from Ce-144 in the stored waste will disappear within 3 to 4 years, and the miscellaneous contributions from short-lived fission products, such as Cs-134, Ru-106, and Pm-147, will decay away within 10 years. Therefore, decay of Sr-90 and Cs-137 would be the major source of heat generation from DWPF waste canisters in a geologic repository.

Currently the contribution of Cs-137 and Sr-90 in fresh waste generated annually corresponds to about 4% of the existing inventory in SRP waste tanks; however, about 2% of this inventory decays each year. Consequently, the heat generation rate of fission products requiring geologic disposal is increasing about 2% annually. Based on current projections of future operations, heat generation from Cs-137 and Sr-90 in stored SRP high-level waste is not expected to exceed 2.0 MW (by the year 2000, the rate of accumulation is expected to be equaled by the rate of decay).

TABLE C-1

Compositional Variations in SRP Waste Sludge

Element*	Amount, wt % Tanks 4 and 6	Tank 5	Tank 13	Tank 15	Tank 16
Fe	32.8	28.9	25.6	5.3	13.9
Al	2.3	1.6	8.7	18.8	16.6
Mn	2.0	5.8	7.8	2.4	2.6
U	9.2	10.8	4.2	3.8	4.5
Na	3.0	5.7	2.6	2.4	2.2
Ca	2.3	0.9	1.8	0.5	2.9
Ni	6.3	6.3	0.4	0.7	0.3

* Present as components with other elements such as oxygen, hydrogen, nitrogen, and sulfur.

Although Pu-238 contributes only 0.5% of the curie content of SRP high-level waste, it will contribute about 8% to the heat generation in canisters. Assuming that the current 8% contribution will continue, the total heat generation in DWPF waste canisters in a geologic repository (containing SRP high-level waste existing and produced over the next two decades) is estimated to be 2.2 MW.

Consequently, if borosilicate glass is selected as the waste form, the average heat generation rate of a DWPF borosilicate glass canister would be about 220 watts based on the production of 10,000 waste canisters. Because of tank-to-tank variations in waste composition, and because of changes in the reference process that may result from ongoing development, the maximum heat generation rate in the DWPF canisters will vary. However, the production techniques can be utilized to limit canister heat generation rates to level within applicable regulatory requirements.

For design purposes (e.g., establishing shielding requirements in the DWPF), the reference DWPF borosilicate glass waste canister is assumed to contain 150,000 Ci of radionuclides and to generate 423 watts.[1] Based upon the projected maximum of 2.2 MW in SRP high-level waste canisters, the average heat generation of about 220 watts per DWPF waste canister will be well below the design basis value and even further below the typical heat rating of canisters containing commercial high-level waste (Table 2.1 of Reference 1).

Calculated surface temperatures of the reference DWPF borosilicate glass waste canister in a salt repository are shown in Figure C-1. The maximum surface temperature occurs approximately 20 years after the waste is emplaced and will be about 80°C in salt[2] and somewhat higher in rock repositories such as granite and basalt. The calculations assume that the canister is generating 256 watts when emplaced in the repository (i.e., 10 years after the reference canister is produced). After the 1000-year containment period (Section 3.2.5.3), waste form surface temperatures would be at ambient repository temperatures; e.g., about 20°C for granite, 35°C for salt, and about 50°C for basalt.

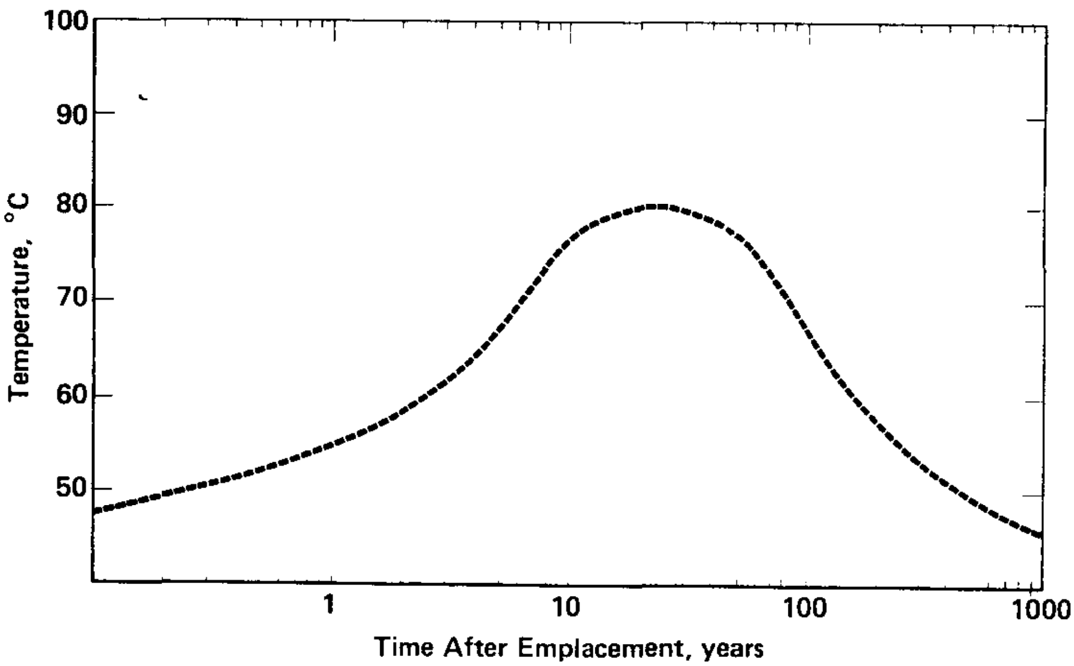

FIGURE C-1. Surface Temperatures for a Reference DWPF Borosilicate Glass Canister in a Salt Repository

REFERENCES FOR APPENDIX C

1. **Final Environmental Impact Statement, Defense Waste Processing Facility, Savannah River Plant, Aiken, SC.** USDOE Report DOE/EIS-0082, U.S. Department of Energy, Washington, DC (February 1982).

2. M. J. Plodinec, G. G. Wicks, and N. E. Bibler. **An Assessment of Savannah River Borosilicate Glass in the Repository Environment.** USDOE Report DP-1629, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, SC (April 1982).